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SUBJECT: Forwards 120-day response to NRC GL 98-04, "Potential for Degradation of ECCS & CSS After LOCA Because of Construction & Protective Coating Deficiencies & Foreign Matl in Containment."

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November 12, 1998

10 CFR 50.54(f)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Ladies/Gentlemen:

Docket Number 50-305
Operating License DPR-43
120-Day Response to NRC Generic Letter 98-04

References: 1. NRC Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, dated July 14, 1998.

2. EPRI TR-109937: Guidelines on the Elements of a Nuclear Safety-Related Coatings Program, dated April 1998.

On July 14, 1998, the Nuclear Regulatory Commission issued the referenced generic letter addressing issues which have generic implications regarding the impact of potential coating debris on the operation of safety-related systems, structures, and components (SSC) during a postulated design basis LOCA. Protective coatings are necessary inside containment to control radioactive contamination and to protect surfaces from erosion and corrosion. Detachment of the coatings from the substrate may make the Emergency Core Cooling System (ECCS) unable to satisfy the requirement of 10 CFR 50.46(b)(5) to provide long-term cooling and may make the safety-related Internal Containment Spray System (ICS) unable to satisfy the plant-specific licensing basis of controlling containment pressure and radioactivity releases following a LOCA. The generic letter requests information under 10 CFR 50.46(f) to evaluate the addressees' programs for ensuring that Service Level 1 protective coatings inside containment do not detach from their substrate during a design basis LOCA and interfere with the operation of the ECCS and the CSS. The NRC intends to use this information to assess whether current regulatory requirements are being correctly implemented and whether these requirements need to be revised.

Attachment 1 provides Wisconsin Public Service Corporation's (WPSC) response to the information requested in NRC Generic Letter 98-04 for the Kewaunee Nuclear Power Plant.

A080

If any additional information is required, please contact David Masarik at (920) 388-8442.

Sincerely,

M L Marchi

Mark L. Marchi
Site Vice President-Kewaunee Plant

DLM

Attach.

cc - US NRC Region III
US NRC Senior Resident Inspector

Subscribed and Sworn to
Before Me This 12th Day
of November 1998

Jeanne M. Ferris
Notary Public, State of Wisconsin

My Commission Expires:
June 13, 1999

ATTACHMENT 1

Letter from Mark L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

November 12, 1998

WPSC 120-Day Response to NRC Generic Letter 98-04

WPSC 120-day Response Generic Letter 98-04

Reqnired Information:

(1) **A snmmary description of the plant-specific program or programs implemented to ensure that Service Level 1 protective coatings nsed inside the containment are procured, applied, and maintained in compliance with applicable regulatory requirements and the plant-specific licensing basis for the facility. Include a discussion of how the plant-specific program meets the applicable criteria of 10 CFR Part 50, Appendix B, as well as iuformation regarding any applicable standards, plant-specific procednres, or other guidance used for: (a) controlling the procnrement of coatings aud paints used at the facility, (b) the qualification testing of protective coatings, and (c) surface preparation, application, surveillance, and maintenance activities for protective coatings. Maintenance activities involve reworking degraded coatings, removing degraded coatings to sonnd coatings, correctly preparing the surfaces, applying new coatings, and verifying the quality of the coatings.**

WPSC RESPONSE

WPSC has implemented controls for the procurement, application, and maintenance of Service Level 1 protective coatings used inside the containment in a manner that is consistent with the licensing basis and regulatory requirement applicable to the Kewaunee Nuclear Power Plant. The requirements of 10 CFR Part 50 Appendix B are implemented through specification of appropriate technical and quality requirements for the Service Level 1 coatings program which includes ongoing maintenance activities.

For the Kewaunee Nuclear Power Plant, Service Level 1[1] coatings are subject to the requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," (June 1973); ANSI N 101.4-1972 "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"; and the Kewaunee Updated Safety Analysis Report (USAR) description. Adequate assurance that the applicable requirements for the procurement, application, inspection, and maintenance are implemented is provided by procedures and programmatic controls, approved under the WPSC Quality Assurance program. WPSC is evaluating the guidance provided in EPRI TR-109937 "Guideline on Nuclear Safety-Related Coatings" and, as appropriate, improvements to our existing programs and procedures for Service Level 1 coatings will be implemented following completion of the evaluation. Completion of this evaluation is scheduled for April 30, 1999.

[1] Our response applies to Service Level 1 coatings used in primary containment that are procured, applied and maintained by Wisconsin Public Service Corporation or their contractor.

Items (a), (b), and (c), discussed below describe the current containment protective coatings program. Item (d) discusses the identification, investigation, and disposition of unqualified containment coatings identified in 1996.

(a) Procurement of Service Level 1 coatings used for new applications or repair/replacement activities are procured in accordance with KNPP Engineering Specification ES-12005 "Protective Coatings for Service Level 1 Applications inside the Reactor Containment Vessel" from a vendor(s) with a quality assurance program meeting the applicable requirements of 10 CFR Part 50 Appendix B. The applicable technical and quality requirements that the vendor is required to meet are specified by WPSC in procurement documents. Acceptance activities are conducted in accordance with procedures that are consistent with ANSI N 45.2 requirements (e.g., receipt inspection, source surveillance, etc.). This specification of required technical and quality requirements combined with appropriate acceptance activities provides adequate assurance that the coatings received meet the requirements of the procurement documents.

(b) The qualification testing of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments referenced above.

(c) The surface preparation, application and surveillance during installation of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meet the applicable portions of the standards and regulatory commitments referenced above. Documentation of completion of these activities is performed consistent with the applicable requirements. Where the requirements of the standards and regulatory commitments did not address or were not applicable to repair/replacement activities, these activities were performed in a manner consistent with the generally accepted practices for coatings repair/replacement. These practices are described in various ASTM standards and coating practice guidelines issued by industry organizations subsequent to those for which WPSC has a regulatory commitment. WPSC recognizes that the NRC has not formally endorsed many of the more recent ASTM standards or industry guidelines, but nonetheless, they provide useful information which can be appropriately applied to provide assurance that repair/replacement activities on Service Level 1 coatings are effective in maintaining the acceptability of the coatings.

(d) In November of 1996 and through discussions with plant personnel, it was identified that unqualified commercial oil-based enamel paints were used in containment on structural and architectural steel, piping, and various equipment. The use of such coatings is contrary to the requirements stated in the original painting specification for containment (X-K177) and Engineering Specification ES-12005 for maintaining the Service Level I coating systems in containment in accordance with Regulatory Guide 1.54 and ANSI N101.4-1972.

The investigation of this issue identified that unqualified coatings were applied to architectural steel (handrails and stairwells) during original construction. Also during a period in the 1980s, additional unqualified enamel based coatings were applied in containment. Two types were predominant, DuPont Dulux enamel and Richardson enamels. The total surface area of applied unqualified coatings in containment was estimated to be approximately 8,430 ft^2. The locations and square footage estimates for each elevation in containment are documented in the Kewaunee corrective action program (Kewaunee Assessment Process (KAP) #471 file). These coatings were found to range in thickness from 2.5 mil up to 18 mil with most coatings in the 3-5 mil range. The material density, based on a sample, is approximately 100 lb/ft^3 for the enamel by itself. The thicker material is enamel over Carboline Carbo-Zinc 11, which is itself a qualified coating. The density of the combined enamel and Carboline is over 230 lb/ft^3 based on a sample. Approximately 40% of the unqualified coatings were removed during the 1996-1997 outage. Analysis has demonstrated that the current amount of unqualified coatings will not affect the operability of the ECCS and ICS following a LOCA.

WPSC periodically conducts condition assessments of containment. When localized coating conditions are identified as degraded, those areas are evaluated and scheduled for repair or replacement, as necessary. The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized. As previously noted, WPSC is evaluating the guidance contained in the EPRI coatings guideline.

(2) Information dcmonstrating compliance with item (I) or item (ii):

(I) For plants with licensing-basis reqnirements for tracking the amount of unqualified coatings inside the containment and for assessing the impact of potential coating debris on the operation of safety-related SSCs during a postulated desigu basis LOCA, the following information shall be provided to demonstrate compliance:

(a) The date and findings of the last assessment of coatings, and the planned date of the next assessment of coatings.

(b) The limit for the amount of nnqnalified protective coatings allowed in the containment and how this limit is determined. Discnss any conservatism in the method used to determiue this limit.

(c) **If a commercial-grade dedication program is being nsed at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the contaiument, discnss how the program adequately qualifies snch a coating for Service Level 1 service. Identify which standards or other guidance are cnrrently being used to dedicate containment coatings at your facility; or,**

WPSC RESPONSE

While KNPP does not have any specific licensing-basis requirements for tracking the amount of unqualified coatings inside the containment, the Kewaunee USAR does specify that the use of unqualified coatings is minimized and that the current amount of unqualified coatings will not affect the operability of the ECCS and ICS following a LOCA. This conclusion is based on two distinct analyses. The first analysis reviewed the location of unqualified coatings and the potential transport velocities to the ECCS recirculation sump. The second analysis assumed transport of 50% of the unqualified coatings to the ECCS recirculation sump and evaluated the effect of the failed coating on the sump inlet screens, ECCS and ICS components including pumps, pump seals, heat exchangers, valves, spray nozzles, flow orifices; reactor core heat transfer and potential blockage, and concluded that operability was not affected.

(ii) For plants without the above licensing-basis reqnirements, information shall be provided to demonstrate compliance with the reqnirements of 10CFR50.46(b)(5), "Long-term cooling" and the functional capability of the safety-related CSS as set forth in your licensing basis. If a licensee can demonstrate this compliauce without quantifyiug the amount of unqualified coatings, this is acceptable.

WPSC RESPONSE:

The following description and referenced materials describe the licensing basis for KNPP relative to conformance with 10 CFR 50.46(b)(5), "Long-term cooling," specifically with regard to KNPP's ability to provide extended decay heat removal including related assumptions for debris that could block containment emergency sump screens.

Updated Safety Analysis Report excerpts on the licensing-basis requirements for KNPP's systems:

6.2.1 DESIGN BASIS

Emergency Core Cooling System Capability

Criterion: An Emergency Core Cooling System with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling systems shall be evaluated conservatively in each area of uncertainty. (GDC 44)

Adequate emergency core cooling is provided by the Emergency Core Cooling System whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection and residual heat removal recirculation.

The primary purpose of the Emergency Core Cooling System is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and ensures that the core will remain substantially intact and in place, with its heat transfer geometry preserved. This protection is afforded for:

a. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b. A loss of coolant associated with the rod ejection accident.

c. A steam generator tube rupture.

To assure effective cooling of the core, limits on peak clad temperature and local metal-water reaction have been established. Results of the multi-rod burst test phase of the Westinghouse Rod Burst Program (Reference 2) show that peak clad temperature calculated during accident increases less than 100°F due to geometry distortion; thus peak clad temperatures determined on the basis of no geometry distortion should be limited to 100°F below 2200°F. However, the peak clad temperature calculated without geometry distortion is limited to 2200°F consistent with the results of the loss of coolant accident analysis...

During the recirculation phase of a loss of coolant accident, the system can accommodate a loss of any part of the flow path since alternative flow path capability is provided...

6.2.2 SYSTEM DESIGN AND OPERATION

Recircnlation Phase

After the injection operation, coolant spilled from the break and water collected from the contaiument spray will be cooled and returned to the Reactor Coolant System by the recirculation paths. The system is arranged so that the residual heat removal pumps take suction from the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is arranged to allow either or both of the residual heat removal pumps to perform the recirculation function. Alternately, the high-head recirculation flow path via the high-head safety injection pumps may be used. This path is only required for the range of small break sizes for which the Reactor Coolant System pressure remains in excess of the shut-off head of the residual heat removal pnmps at the end of the injection phase.

During recirculation, one recirculation train will be in service which includes either of the two residual heat removal pumps and the associated residual heat exchanger. The flow will go from the discharge of the residual heat removal pump through the residual heat exchanger and then into the reactor via either a low-head injection path or a high-head injection path via a safety injection pump. The high-head injection paths are provided in the event of a small break in which the pressure in the Reactor Coolant System is higher than the shutoff head of the residual heat removal pump.

In the event of a failure in the operating train during recirculation, the capability exists to switch to the other independent recirculation flow path. The design of the containment drains are shown in Figures 6.2-3 and 6.2-4. As illustrated, the containment building serves as a sump, after the stored water (up to 270,000 gal.) has been injected and the recirculation phase begins.

Foreign matter is prevented from entering the recirculation system by two screens mounted over the sump inlet. These screens are conical in shape, manufactured of Johnson Screen material and sized to prevent any particles larger than 1/8 inch from entering the sump.[2]

The NRC originally accepted these analyses and these systems as meeting the requirements of 10 CFR 50.46(b)(5) in Safety Evaluation of the Kewaunee Nuclear Power Plant, Docket No. 50-305, issued July 24, 1972. The following material is excerpted from the Safety Evaluation:

6.3 Emergency Core Cooling

The emergency core cooling system (ECCS) for this plant consists of the same three basic subsystems we reviewed and found acceptable for Ginna, H. B. Robinson, and Point Beach. These subsystems are (a) the high-head injection system (SIS), (b) the low-head injection system (RHR), and (c) the accumulators....

On June 29, 1971, the AEC issued an Interim Policy Statement[3] containing interim acceptance criteria for the performance of emergency core cooling systems for light-water cooled nuclear power reactors....

In Appendix F to the FSAR the applicant provided the results of an analysis of the ECCS performance capability for the Plant. This analysis was performed using the Westinghouse evaluation model described in Appendix A, Part 3 of the Interim Policy Statement.

In our review of the applicant's analysis, we first considered the events that occur during the blowdown period, defined as the time from occurrence of the postulated pipe break to the time that the reactor coolant system pressure is reduced to the containment pressure. The second phase of the postulated accident, called the refill and reflood period, starts at

[2] KAP 0885 has previously reviewed this USAR statement and concluded that 1/8 inch value is imprecise, since screen openings are actually 1/8" x 15/32".

[3] 36 Federal Register, 12247

> the end of blowdown and stops when the temperature transient of the fuel cladding is satisfactorily controlled. To meet the acceptance criterion of the Interim Policy Statement limiting the calculated peak clad temperature to less than 2300°F, the applicant proposes to operate the nuclear unit in such a manner that a peak linear power density of 16.8 kW/ft at 100% power is not exceeded. This Operating restriction is imposed through the Technical Specifications by limiting the maximum allowable peaking factors.
>
> In Appendix F to the FSAR, the applicant also presented the results of analyses of calculated peak clad temperatures for a spectrum of pipe break sizes up to and including the double-ended rupture of the largest coolant pipe...
>
> The fuel cladding remains at an elevated temperature for only a short period of time. No significant amount of cladding would become embrittled and the core geometry would be preserved. As a result, the long-term removal of decay heat would be carried out effectively by the emergency core cooling system...
>
> We have reviewed the applicant's analysis of the consequences of small breaks requiring the operation of the emergency core cooling system. The peak clad temperatures calculated by the applicant for this class of breaks is less than 1500°F. While we are in the process of performing a more detailed review of small break analyses for PWR plants on a generic basis, we have concluded that the information presently available on the Kewaunee application provides adequate assurance of acceptable ECCS performance for the small break class of accidents.
>
> The calculated maximum fuel element cladding temperature of 2010°F is well under the 2300 limit stated in the interim acceptance criteria. The total core metal-water reaction calculated to occur as a result of the maximum temperature break is less than 0.1% which is well under the 1.0% criterion for interim acceptance. The clad temperature transient is terminated while the core is still amenable to cooling and before it becomes excessively embrittled, such that its essential heat transfer geometry is preserved and it can be cooled to remove decay heat for an extended period of time. We conclude that the ECCS design for Kewaunee meets the interim acceptance criteria for ECCS performance.

The ECCS Analysis was subsequently updated to address the final 10 CFR 50.46 and has been revised on several occasions to address industry issues and plant modifications. The conclusions of the current small break analysis of record are that: "The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided."[4]

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[4] WCAP-14103, "Westinghouse Small Break Loss-of-Coolant Accident, Kewaunee NOTRUMP Analysis," June 1994

Similar conclusions have been drawn from the large break LOCA ECCS analysis: "The large break LOCA analysis... shows that the high head and low head safety injection, together with the accumulators, provide sufficient core flooding to meet the 10 CFR 50.46 Acceptance Criteria...The core remains amenable to cooling.... The core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required for the long-lived radioactivity remaining in the core."

The licensing basis for KNPP, as described in the ECCS analysis and as accepted by the NRC's SERs, provides both the regulatory and safety basis for safety system performance. Coatings are not treated separately in the licensing basis for KNPP, because the sump screen blockage assumption does not distinguish among the source items for the LOCA-generated debris. As the NRC noted in NRC Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," a change in regulatory guidance for the basis for sump screen blockage would constitute a generic backfit. Moreover, the analysis for coating failure alone during a LOCA, and testing of coating failure conducted to date, does not contradict the KNPP determination that emergency core cooling system flow following a LOCA will be adequate to maintain the core temperature at an acceptable low value and to remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core following a design-basis accident. Accordingly, a separate demonstration of the regulatory and safety basis for safety system performance is not required.

The following information sball be provided:

(a) If commercial-grade coatings are being used at your facility for Service Level 1 applications, and such coatings are not dedicated or controlled under your Appendix B Quality Assurance Program, provide the regulatory and safety basis for not controlling these coatings in accordance with such a program. Additionally, explain why the facility's licensing basis does not require such a program.

WPSC RESPONSE

WPSC does not currently use commercial grade dedication for Service Level 1 coatings used inside containment at KNPP.